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THE ROLE OF A BLANKET TRITIUM SYSTEM ON THE FUSION FUEL CYCLE

D.K. SZE, P. FINN, R. CLEMMER
Argonne National Laboratory, 9700 S, Cass Ave., Argonne, IL 60439, USA

J. ANDERSON, J. BARTLIT
Los Alamos National Laboratory, PO Box 1663, Los Alamos NM 87545, USA

Y. NARUSE and H. YOSHIDA
Japan Atomic Energy Research Institute, Japan

The requirements of tritium technology are centered in three main areas, i.e., (1) fuel processing, (2) breeder tritium extraction, and (3) tritium containment. The gaseous tritium stream from the breeder tritium extraction system is significantly different from the plasma exhaust stream and, therefore, may have an important impact on the operation of the fuel processing system. For some blankets, such an aqueous solution blanket, the blanket tritium stream may dominate the fuel processing system in terms of component size and power consumption. The importance of the blanket interface to a fuel processing experiment, such as TSTA, has been identified. The initial work to define the blanket processing system, which is proposed to be added as part of TSTA, will be discussed here.

1. Introduction

The requirements of tritium technology are centered in three main areas, (1) fuel processing, (2) breeder tritium extraction, and (3) tritium containment. The Tritium Systems Test Assembly (TSTA) [1,2] now in operation at Los Alamos National Laboratory (LANL) is dedicated to developing and demonstrating the tritium technology for fuel processing and containment. TSTA is the only fusion fuel processing facility, in existence or in planning, that can operate in a continuous closed-loop mode. The tritium throughput of TSTA is 1000 g/d. However, TSTA does not have a blanket interface system. The initial TSTA proposal of 1976 included a blanket interface, as part of TSTA, to simulate the processing of the tritium gas stream from the blanket extraction system. However, due to a limited budget and limited information on blanket systems at that time, the blanket interface was not included.

Blanket processing is an important part of the fusion fuel cycle. It is the interface between the blanket tritium extraction system and the fuel processing system. Although the tritium throughput in the plasma exhaust is a factor of 20 to 100 higher than in the blanket system (corresponding to a plasma burn fraction of 1–5%), the total hydrogen isotope throughput may be dominated by the blanket stream. For the aqueous solution blanket, a candidate for both US ITER and NET fusion reactor designs, the hydrogen throughput in the blanket tritium system is a factor of 1000 higher than in the plasma exhaust. In addition, the helium content, impurity composition, and radionuclides in the blanket stream are very different from those in the plasma exhaust. The goal of the TSTA experiments is to demonstrate that the fusion fuel cycle can be operated under the constraints of engineering, environmental safety, and economics. Therefore, to develop and demonstrate tritium technology for fuel processing, it is important to add blanket processing to TSTA.

Blanket tritium recovery concepts have been developed as part of the fusion reactor design studies from various forms of breeding material. Selected experiments have also been carried out to resolve key issues to verify the blanket tritium recovery concept. However, almost all the work performed so far is aimed primarily to minimize the blanket tritium inventory and, to a lesser degree, to minimize the tritium leakage rate. Relatively little effort has been devoted to investigating the problems associated with fuel processing. Therefore, the link between the blanket tritium recovery system and

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the fuel processing system is missing. To demonstrate the complete fusion fuel cycle, it is important to define, design, and operate the blanket processing system mockup as part of TSTA. The blanket processing system mockup will simulate the gaseous tritium from the blanket tritium recovery system and the required cleanup steps before it will be combined with the plasma exhaust stream.

To design a blanket processing system mockup, the condition of blanket tritium streams (impurities, H/T ratio, radionuclides, helium content, corrosive species, etc) for various fusion reactor blankets have to be defined. The information on the possible ranges of conditions that may be encountered in blanket tritium streams is used to define the processing requirements and formulate a conceptual design of the blanket tritium processing system. A key part of this effort is the evaluation of the interface between the blanket processing system and the rest of the tritium cycle. This means that the points where the stream could be handled by the TSTA (e.g., at the inlet of the Fuel Cleanup Unit [FCU] or the inlet of the Isotope Separation System [ISS], or possibly at other points in the TSTA system) will be evaluated. The specific goal is to identify which operations (purification, isotope enrichment, etc) require special handling for the blanket streams.

In support of this effort, the characteristics of the blanket tritium steams from one type of blanket were evaluated: namely, self-cooled liquid lithium. Later phases of the study will investigate other blanket concepts, including Li_2O – a solid breeder blanket and an aqueous solution blanket.

2. Status of TSTA

There are currently several existing tritium laboratories/facilities dedicated to fusion tritium technology. The major facilities in the US include the TSTA and the Tritium Plasma Experiment (TPX) now in operation. The Tritium Processing Laboratory (TPL) has been constructed at JAERI in Japan. The TPL will begin tritium operations in 1987. In Europe the French CEA Laboratory at Bruyere-le-Chatél has now started to do fusion tritium experiments. At Karlsruhe (KfK) a tritium laboratory, which will emphasize work on fuel processing systems and components and blanket tritium recovery systems, will soon be constructed, with first tritium tests in 1989–1990. At Ispra the Joint Research Centre is constructing a tritium laboratory for safety studies. Some work on blanket extraction systems will

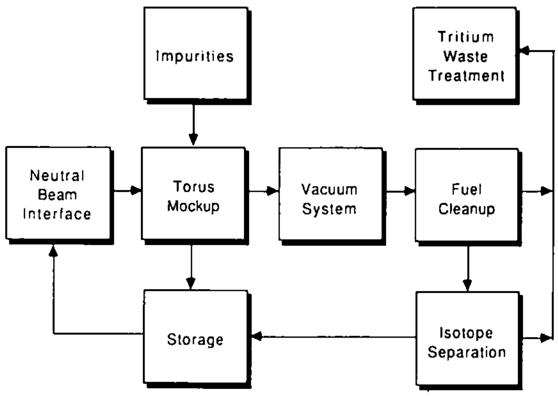

Fig. 1. TSTA present configuration.

Table 1
Comparison of fusion fuel cycle facilities

	US (TSTA)	European countries			Japan	USSR
		W. Germany (KfK)	Italy (Ispra)	France	JAERI	
Experience	Defense program extrapolated into fusion	No prior experience	No prior experience	Defense program extrapolated into fusion	No prior experience	Assumed capability from defence prog.
Nature of facility	Complete plasma exhaust cycle	Component test	Component test	Unknown	Individual process test	Unknown
Facility size	INTOR size (T inventory 100 g)	1/10 INTOR (T inventory 10 g)	1/10 INTOR (T inventory 10 g)	Unknown	1/10 INTOR (T inventory 10 g)	Unknown
Tritium exp. status date	June 1984	~1990	~1990	Unknown	~1987)	Unknown

Table 2
Facilities to be used in tritium-processing development

Tritium systems test assembly (TSTA) with blanket-development facility.	
Purpose:	Blanket-product processing integration.
Features:	Required processing technology (i.e., extraction processes, perhaps additional isotope separation). Comparable in scale to TSTA. This interface could be added either at TSTA or at the blanket facility.
Cost:	Facility revision – about $5 million; the annual operating cost for the multipurpose process-development program would be about $3 million/y for perhaps an added 2.5 y.
Date required:	1997

be done at the Ispra tritium facility. This laboratory will begin tritium testing in 1989–1990. In Canada the Canadian Fusion Fuel Technology Project is considering the construction of a new experimental laboratory. This laboratory will complement the tritium recovery systems being built by Ontario Hydro at Darlington and the AECL at Chalk River.

A comparison of the major fusion fuel cycle facilities is shown in table 1 TSTA is not also the only facilitiy operating with tritium for developing fusion fuel processing, but also the only integrated facility, in existence or planned, to operate a continuous flow processing of fusion fuels. TSTA has processed tritium for five continuous days at the rate of 1000 g/d, with a tritium inventory of 100 g. This operation demonstrates fusion fuel processing in the order of the ITER scale.

The basic TSTA flow diagram is shown in fig. 1. As can be seen from this figure, the role of TSTA as it stands today is to simulate the processing of the plasma exhaust. The TSTA initially proposed [2], the TSTA fabrication design [1], the FINESSE study [3], and the TPA study [4], all recognize the importance of the blanket interface to the demonstration of the fusion fuel cycle. Table 2 shows the recommendation of TPA to incorporate the "Blanket Development Facility" and TSTA. Mainly due to budgetary reasons, blanket operation has not been a part of TSTA operation.

3. Role of blanket interface

The proposed blanket interface is to join a blanket processing system mockup to that of the existing TSTA. This upgraded TSTA system can more closely simulate the entire fusion fuel procdess. The flow diagram of TSTA with the addition of the blanket processing system mockup is shown in fig. 2. The assumptions and sealing of the blanket processing mockup should be similar to that of TSTA. There are three major components in the blanket processing mockup, i.e.,

(1) *The blanket stream mockup*. The function of this component is identical to that of the torus mockup. It is a mixing tank in which a predicted blanket gaseous tritium stream is prepared.
(2) *Tritium cleanup system*. This component is to process the blanket tritium stream to a condition acceptable by TSTA.
(3) *Blanket interface*. This component is required to treat the blanket tritium stream further to be compatible to the location of the interface. A potential interface will be a cryogenic distallation column to increase the T/D/H ratio to be acceptable by the CD in TSTA.

Although the tritium flow rate in the plasma exhaust exceeds that in the blanket stream by about a factor of 10, the hydrogen isotopes throughput may very well be dominated by the blanket system. For the helium cooled solid breeder blanket and aqueous solution blanket, the H/T ratio in the blanket can range between 10^3 to 10^5.

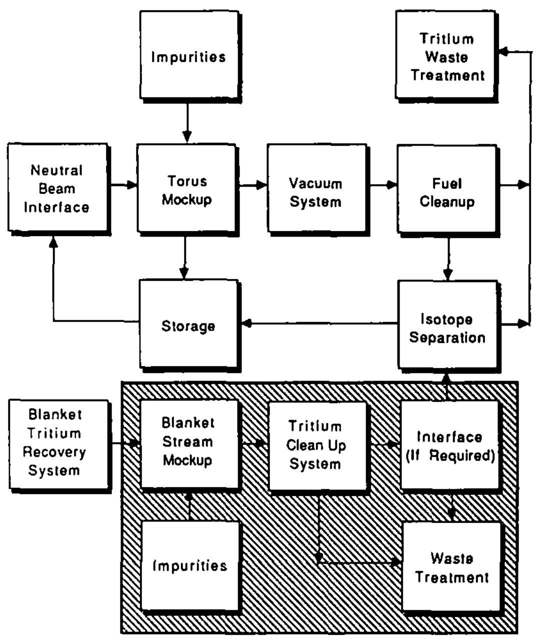

Fig. 2. Upgraded TSTA. (With blanket processing system mockup).

Table 3
Comparison of the tritium fuel streams from blanket and plasma

	Blanket	Plasma
$(H+D)/T$	up to 10^5	1
T/day	50 g	1 kg
Total $(H+D+T)$/day	5000 kg	2 kg
Impurity sources	Li Chemistry, H_2O, CO_2, halides, water chemistry	Air, Structural material, fusion reaction
Impurities	$LiOH$, CO_2 halides, acids, radionuclides	H_2O, N_2, He^3, CH_x, Ar, NH_x, radionuclides

Therefore, the hydrogen isotopes throughput in the blanket stream dominates by a factor to 10^2 to 10^4. Therefore, it is important to include the blanket tritium stream if a tritium fuel cycle is to be demonstrated in TSTA. In addition, the nature of impurities and their corrosive behavior is different between the plasma exhaust and the blanket systems. The blanket operating conditions may be more stringent due to the high temperature, corrosive species, presence of radionuclides, and pressure-flow transients. Table 3 compares the conditions of the tritium fuel streams from the blanket and the plasma. The role of the blanket interface is to ensure that two such different streams can be combined and operated within one system.

The initial steps required to define the blanket interface is to:
(1) Select a tritium recovery method for a blanket.
(2) Define the conditions and compositions of the tritium stream from the blanket recovery system.
(3) Define a chemistry process to cleanup the tritium stream which is acceptable by TSTA requirements.
(4) Define the condition and location of the interface.
This study will be a conceptual design. Actual experimental work will be discussed in the future.

4. Initial work

The characteristics of the blanket tritium streams from one type of blanket have been evaluated: namely, the self-cooled liquid lithium [5]. A brief summary of the work will be discussed here. The work includes:
(1) An assessment of the options for tritium recovery from liquid lithium and the selection of the reference method (molten salt extraction).
(2) A definition of reference blanket, salt, and sparge stream conditions.
(3) An evaluation of the source terms in each stream including activations of impurities.
(4) The determination of the condition of the blanket tritium gaseous stream.

Similar investigations will be performed for other blanket concepts, including the helium-cooled, helium purged lithium oxide blanket, and a self-cooled aqueous solution blanket. The purpose of the evaluation is for ITER applications.

The sparge gas coming out of the electrolysis unit is the blanket tritium product stream that is to be sent to the blanket processing system. The characteristics of this stream are summarized in table 4. The major features of this stream are summarized below:

(1) The stream is helium gas, plus the tritium product, plus impurities. The concentration of hydrogen isotopes in the stream is about 1000 vppm, or 100 Pa. These products must be extracted from the tritium under conditions where the helium is separated from the product to a high degree. The blanket stream is 99.9% helium, compared to the plasma exhaust stream, which is about 5% helium.

(2) The ratio of H to T in the product stream is about one. The implication of this is that the blanket product stream for a liquid lithium blanket may not require its own isotopic separation system. This is contrasted to other blanket concepts where H/T ratios may be as high as 10^3 to 10^5.

(3) The amount of deuterium in the product stream is very small, about 0.3% of the tritium level.

(4) A significant level of nitrogen gas may be present in the sparge stream; levels were calculated to about 2000 ppm. This gas will have to be separated from the product stream.

Table 4
Conditions of sparge gas

Sparge gas	He
Sparge gas flow rate	66 000 l/h
Temperature	500 °C
Pressure	0.1 MPa (1 atm)
T_2 level, vppm	816.0
HT level, vppm	490.0
H_2 level, vppm	294.0
$D_2(HD+DT)$ level, vppm	3.0
N_2 level, vppm	2200.0
NH_3 level, vppm	0.020
LiBr level, vppm	2.37
LiCl level, vppm	0.03
LiF level, vppm	< 600.0?
PH_3	?
SiH_x	?

(5) Owing to the vapor pressure of halides at 500 °C and to entrainment, there will be lithium halides present at levels of about 1 vppm in the sparge gas. These species must be removed to prevent corrosion of other processing components.

(6) The product stream is expected to contain small amount of beta emitters, including C-14 and possibly P-32 and S-35.

JANAF thermochemical data [6] were used to calculate equilibrium amounts of NH_3 and the vapor pressures of the lithium halides. In addition to these species, a significant amount of acetylene could be present. The amount is uncertain because of a lack of data on electrolysis behavior. The acetylene will contain about 0.06 mCi of C-14 per milligram of C_2HT, as a result of nitrogen activation. This compares to a tritium activity of 1.0 Ci per milligram of C_2HT. There are likely to be trace levels of a number of radionuclides in the purge gas, but the analysis to date has not identified a pathway for any gamma-emitters to be present at significant levels.

5. Conclusions

The conditions for the gaseous tritium stream from the tritium recovery system of a self-cooled, lithium blanket have been evaluated. The total throughput (dominated by helium), the hydrogen-to-tritium ratio, the corrosion-product composition (mainly halides), and the activation products are all defined from the best available information. For a liquid lithium blanket, the results show several factors which must be addressed in the design of a mockup blanket exhaust processing system:

(1) The total gas throughput in the blanket system is much larger than that of the TSTA (1000 mol/h versus 15 mol/h). The blanket throughput is dominated by the 10^3 excess of helium sparge gas. Separation of the hydrogen isotopes from helium is feasible However, the system to be used needs to be defined.

(2) The ratio of protium to tritium in the blanket stream is about one. This ratio may be much higher for a very short time at startup, owing to residues of H in the starting Li and HT-9. A study to show the trade-off between the use of a common cryogenic distillation unit with the plasma exhaust and the use of a separate unit to avoid mixing H, D, and T needs to be done.

(3) The composition and amount of corrosive chemicals in the blanket stream is dominated by the halides. A method to remove the halides to an acceptable concentration is needed.

(4) The major radionuclides found were C-14 from nitrogen and P-32 from chlorine. A method to remove the radionuclides to an acceptable concentration, to be defined, is needed.

(5) The nitrogen level expected in the blanket stream is much higher than that anticipated for the TSTA plasma exhaust system. A method of removing nitrogen to an acceptable level is needed.

The blanket processing system mockup for TSTA for a sefl-cooled lithium blanket will incorporate units that can address these issues. The exact configuration of this processing system will be defined in the next stage of this program, as will the interface with TSTA.

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